

Constellation Energy

Nine Mile Point Nuclear Station

P.O. Box 63
Lycoming, NY 13093

May 6, 2005
NMP1L 1948

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

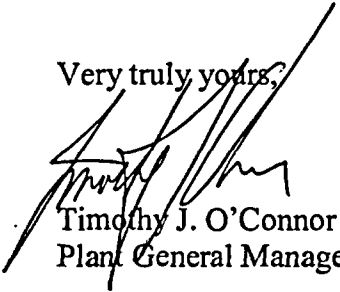
SUBJECT: Nine Mile Point Unit 1
Docket No. 50-220
Facility Operating License No. DPR-63

Licensee Event Report 05-001, "Automatic Reactor Scram due to a Failure in the Circuitry for a Moisture Separator Tank Level Switch"

Gentlemen:

In accordance with 10 CFR 50.73(a)(2)(iv)(A), we are submitting Licensee Event Report 05-001, "Automatic Reactor Scram due to a Failure in the Circuitry for a Moisture Separator Tank Level Switch."

Very truly yours,


Timothy J. O'Connor
Plant General Manager

TJO/KSE/sc
Attachment

cc: Mr. S. J. Collins, NRC Regional Administrator, Region I
Mr. G. K. Hunegs, NRC Senior Resident Inspector

IE22

LICENSEE EVENT REPORT (LER)

(See reverse for required number of
digits/characters for each block)

Estimated burden per response to comply with this mandatory collection request: 50 hours. Reported lessons learned are incorporated into the licensing process and fed back to industry. Send comments regarding burden estimate to the Records and FOIA/Privacy Service Branch (T-5 F52), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to infocollects@nrc.gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202, (3150-0104), Office of Management and Budget, Washington, DC 20503. If a means used to impose an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection.

1. FACILITY NAME Nine Mile Point, Unit 1	2. DOCKET NUMBER 05000220	3. PAGE 1 OF 4
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4. TITLE Automatic Reactor Scram due to a Failure in the Circuitry for a Moisture Separator Tank Level Switch
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5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
03	07	2005	2005	- 001 -	00	05	06	2005	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE 1	11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR§: (Check all that apply)									
10. POWER LEVEL 100	<input type="checkbox"/> 20.2201(b)	<input type="checkbox"/> 20.2203(a)(3)(i)	<input type="checkbox"/> 50.73(a)(2)(i)(C)	<input type="checkbox"/> 50.73(a)(2)(vii)						
	<input type="checkbox"/> 20.2201(d)	<input type="checkbox"/> 20.2203(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(ii)(A)	<input type="checkbox"/> 50.73(a)(2)(viii)(A)						
	<input type="checkbox"/> 20.2203(a)(1)	<input type="checkbox"/> 20.2203(a)(4)	<input type="checkbox"/> 50.73(a)(2)(ii)(B)	<input type="checkbox"/> 50.73(a)(2)(viii)(B)						
	<input type="checkbox"/> 20.2203(a)(2)(i)	<input type="checkbox"/> 50.36(c)(1)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(iii)	<input type="checkbox"/> 50.73(a)(2)(ix)(A)						
	<input type="checkbox"/> 20.2203(a)(2)(ii)	<input type="checkbox"/> 50.36(c)(1)(ii)(A)	<input checked="" type="checkbox"/> 50.73(a)(2)(iv)(A)	<input type="checkbox"/> 50.73(a)(2)(x)						
	<input type="checkbox"/> 20.2203(a)(2)(iii)	<input type="checkbox"/> 50.36(c)(2)	<input type="checkbox"/> 50.73(a)(2)(v)(A)	<input type="checkbox"/> 73.71(a)(4)						
	<input type="checkbox"/> 20.2203(a)(2)(iv)	<input type="checkbox"/> 50.46(a)(3)(ii)	<input type="checkbox"/> 50.73(a)(2)(v)(B)	<input type="checkbox"/> 73.71(a)(5)						
	<input type="checkbox"/> 20.2203(a)(2)(v)	<input type="checkbox"/> 50.73(a)(2)(i)(A)	<input type="checkbox"/> 50.73(a)(2)(v)(C)	<input type="checkbox"/> OTHER						
<input type="checkbox"/> 20.2203(a)(2)(vi)	<input type="checkbox"/> 50.73(a)(2)(i)(B)	<input type="checkbox"/> 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A							

12. LICENSEE CONTACT FOR THIS LER									
NAME M. Steven Leonard, General Supervisor Licensing								TELEPHONE NUMBER (include Area Code) 315-349-4039	

#13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT									
CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
E	SB	EB	GE	Yes					

14. SUPPLEMENTAL REPORT EXPECTED					15. EXPECTED SUBMISSION DATE		MONTH	DAY	YEAR
<input type="checkbox"/> YES (If yes, complete 15. EXPECTED SUBMISSION DATE)					<input checked="" type="checkbox"/> NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On March 7, 2005, at approximately 0437 hours, with the mode switch in "RUN" and reactor thermal power at approximately 100%, Nine Mile Point Unit 1 experienced an automatic reactor scram initiated by a turbine trip caused by a failure in the circuitry of a Moisture Separator Tank level switch.

A spurious actuation of the Moisture Separator Tank level switch was caused by a short circuit between two terminal points caused by water intrusion into the associated junction box. Based on inspections performed, the most probable source of the water inside the junction box was condensation of steam from a packing leak in an instrument root valve in the near vicinity. Corrosion products accumulated at adjacent terminal points due to previous water intrusions which created moisture conditions over an extended period of time leading to a short circuit.

To prevent recurrence, the terminal points were cleaned and all top and side external penetrations of the junction box were sealed. A design change to the trip logic of the Moisture Separator Tank level switches to prevent a trip on a single level switch contact failure was installed and tested in refuel outage 18. Inspections and repairs were performed in refuel outage 18 on a defined population of junction boxes with similar environmental conditions.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

I. Description of Event

On March 7, 2005, at approximately 0437 hours, with the mode switch in "RUN" and reactor thermal power at approximately 100%, Nine Mile Point Unit 1 experienced an automatic reactor scram initiated by a turbine trip. The turbine trip was initiated by a high level signal from a Moisture Separator Tank (122). At the time of the scram, no maintenance, testing, or plant evolutions related to the Moisture Separator Tank, its Instrumentation, or its support systems were in progress.

All control rods fully inserted. Reactor water level lowered as expected following the scram. As designed, High Pressure Coolant Injection (HPCI) initiated automatically as a result of the turbine trip. As designed, the High Reactor Water Level signal (95 inches) tripped the feedwater pumps and reactor water level was restored to the required band. Operators stabilized reactor pressure and level and transitioned out of the Emergency Operating Procedures (EOPs) into normal operating procedures. Reactor cool down was commenced at 0558 hours.

II. Cause of Event

The direct cause of the turbine trip and the resulting reactor scram was water intrusion into a junction box (19721-Y) associated with wiring from the level switch (LS-24-28) monitoring Moisture Separator Tank (122) level. Spurious actuation of the Moisture Separator Tank level switch was caused by an electrical short circuit between two terminal points in the junction box.

Based on inspections performed, the most probable source of the water inside the junction box was condensation of steam from a past packing leak in an instrument root valve in the near vicinity. Although there was not an active leak, corrosion products accumulated at adjacent terminal points due to previous water intrusions which created moisture conditions over an extended period of time leading to a short circuit.

III. Analysis of Event

As designed, High Pressure Coolant Injection (HPCI) initiated automatically as a result of the turbine trip. At the time of the scram, 12 feedwater pump was operating in manual and 11 feedwater pump was in standby. Upon HPCI initiation, feedwater pump 11 started and was available for injection; reactor water level was restored from 31.6 inches (the lowest level reached during the transient). Feedwater pump 13 was secured and HPCI logic was reset. The High Reactor Water Level signal (95 inches) tripped the feedwater pumps, as designed, and reactor water level was restored to the required band. Reject flow to the main condenser was initiated before reaching the High Reactor Water Level signal. The highest reactor water level reached during the transient was 96.3 inches which is below the Emergency Condenser (EC) and Main Steam line nozzles.

The Emergency Operating Procedures (EOPs) were entered for RPV Control on reactor water level less than 53 inches. When conditions stabilized, the EOPs were exited and normal operating procedures were used to continue the plant shutdown. The reactor cool down was commenced at 0558 hours. The reactor cooldown rate was maintained less than 75 degrees F/hr until the reactor reached Cold Shutdown in accordance with Technical Specifications.

As expected, reactor pressure rose from an initial pressure of 1022 psig before the scram and reached approximately 1072 psig during the event. Three electromatic relief valves (113, 122 and 123) opened momentarily, as designed, to lower pressure. Operators used the turbine bypass valves to control reactor pressure during plant cooldown in accordance with operating procedures. No abnormalities were encountered in reactor pressure control.

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III. Analysis of Event (Continued)

As designed, house electrical loads auto transferred to the reserve supply following the scram. No adverse or unexpected electrical transients occurred. No automatic start signals were initiated or expected for the emergency diesel generators during this event.

No systems or components were inoperable at the start of the event that contributed to the severity of the event. All plant safety systems and automatic initiations responded as expected.

Based on a probabilistic risk assessment (PRA), a turbine trip from full power with all risk significant equipment/systems available results in a Conditional Core Damage Probability (CCDP) of less than 1E-6. Therefore this event is considered to have low risk significance. Based on the above, the event did not pose a threat to the health and safety of the public or plant personnel.

An NRC 10 CFR 50.72 report (Event Number 41464) was made on March 7, 2005 at 0600 hours to report the scram and HPCI initiation. The Emergency Plan was not activated because no entry conditions were met.

This event is reportable in accordance with 10 CFR 50.73(a)(2)(iv)(A) as an event or condition that resulted in manual or automatic actuation of the Reactor Protection System. The automatic initiation of HPCI is also reportable under 10 CFR 50.73(a)(2)(iv)(A).

IV. Corrective Actions

A. Action Taken to Return Affected Systems to Pre-Event Normal Status:

- Junction Box 19721-Y was dried, corroded and degraded terminal lugs were replaced, terminal points were cleaned, and the junction box was sealed to prevent future water intrusion.

B. Action Taken or Planned to Prevent Recurrence:

NOTE: There are no NRC regulatory commitments in this Licensee Event Report.

- An inspection program was established to determine extent of condition. Based on that review, five other junction boxes were identified where water intrusion due to dripping or impinging water could cause a short circuit that would result in a plant trip. These junction boxes were inspected, repaired and sealed during refuel outage 18 to prevent similar events.
- A design change was implemented to add redundant trip logic to the moisture separator tank level switches as part of Single Point Vulnerability Reduction Program. This was completed and tested during refuel outage 18.
- The Fluid Leak Management Program will be validated and revised as necessary to assure that proper investigation and evaluation of risk significance is determined upon leak discovery.
- For Nine Mile Point Unit 2, during the next scheduled refuel outage, inspections of junction boxes that may have been subjected to water intrusion will be completed to identify signs of degraded electrical connections.

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NARRATIVE (If more space is required, use additional copies of NRC Form 366A) (17)

V. Additional Information

A. Failed Components:

Junction Box, Terminal Strip

B. Previous Similar Events:

A similar LER event historical search was conducted with the following results: No documentation of similar LER events with the same root cause at Nine Mile Point Nuclear Station Unit 1 could be identified.

C. Identification of systems and components referred to in this Licensee Event Report:

<u>Components</u>	<u>IEEE 805 System ID</u>	<u>IEEE 803A Function</u>
Main/Reheat Steam System	SB	N/A
Reactor Recirculation System	AD	N/A
High Pressure Coolant Injection System	BJ	N/A
Electromatic Relief Valves	SB	PSV
Control Rods	AA	ROD
Reactor Protection System	JC	N/A
Turbine Bypass Valves	SB	PCV
Emergency Diesel Generators	EK	DG
Level Switch	SB	LS
Junction Box	SB	JBX
Terminal Strip	SB	EB
Pump	SJ, AD	P
Moisture Separator Tank	SB	TK
Motor Generator Set	AD	MG
Main Condenser	SG	COND